NON-PUBLIC?: N
ACCESSION #: 9209300077
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 OF 06

DOCKET NUMBER: 05000410

TITLE: Reactor Scram on Low Reactor Water Level Caused by a Loss of
Feedwater Pumps to Personnel error
EVENT DATE: 08/22/92 LER #: 92-017-00 REPORT DATE: 09/21/92

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 055

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Mr. Alan DeGracia, Manager TELEPHONE: (315) 349-7531
Operations NMP2

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On August 22, 1992 at 0333 hours, with the reactor mode switch in the "RUN" position and the plant operating at approximately 55 percent rated thermal power, Nine Mile Point Unit 2 (NMP2) experienced a reactor scram on a
reactor vessel low water level (Level 3) signal. Specifically,
while a planned shift of Feedwater System (FWS) pumps was in progress to allow maintenance activities on the FWS, a degraded condition on the Condensate System (CNM) occurred, resulting in a loss of feedwater flow to the reactor. Reactor vessel water level lowered to 159.3 inches (Level 3 trip setpoint), initiating an automatic reactor scram signal, and a Group 4 Primary Containment isolation (Residual Heat Removal sample and discharge isolation valves).

The root cause for the reactor scram was personnel error.

Immediate actions included restoring reactor vessel inventory and commencing a controlled plant shutdown. Additional corrective actions include: 1) Short term training of operators; 2) long term changes to the operator training program; 3) issuing a Lessons Learned Transmittal; 4) counseling on-shift personnel; 5) evaluating FWS operation; and 6) revising Operating Procedures.

END OF ABSTRACT

I. DESCRIPTION OF EVENT

On August 22, 1992 at 0333 hours, with the reactor mode switch in the "RUN" position and the plant operating at approximately 55 percent rated thermal power, Nine Mile Point Unit 2 (NMP2) experienced a reactor scram on a reactor vessel low water level (Level 3) signal. Specifically, while a planned shift of Feedwater System (FWS) pumps was in progress to allow maintenance activities on the FWS, a degraded condition on the Condensate System (CNM) occurred, resulting in a loss of feedwater flow to the reactor. Reactor vessel water level lowered to 1 5 9.3 inches (Level 3 trip setpoint), initiating an automatic reactor scram signal, and a Group 4 Primary Containment isolation (Residual Heat Removal sample and discharge isolation valves).

During the 2200-0600 shift on August 22, 1992, a planned rotation of operating Feedwater pumps was in progress to allow maintenance on the seals for Feedwater System pump 2FWS-P1C. Reactor power had been lowered to approximately 55 percent from 100 percent rated thermal power to support the Feedwater pump shift evolution in accordance with Operating Procedure N2-OP-3, "Condensate and Feedwater System." Additionally, prior to the Feedwater pump rotation, the Condensate and Feedwater pumps were operating in a 3-3-2 configuration (3 Condensate pumps, 3 Condensate Booster pumps, and 2 Feedwater pumps).

Before commencing the pump shift evolution, Control Room operators held a briefing to discuss the evolution and decided to monitor the Condensate a nd Feedwater System parameters closely both before and during the evolution.

The sequence of events leading to the Level 3 signal and subsequent reactor scram was as follows:

August 22, 1992

o At 2305 hours, the operators commenced a reactor power reduction from 100 to 55 percent in order to shift Feedwater pumps. This was done in accordance with N2-OP-3, Section 4.3, "Shifting FWS pumps by starting the third and then taking off the desired pump."

August 23, 1992

o At 0304 hours, reactor power was 55 percent of rated thermal power.

o At 0310 hours, operators started Feedwater pump 2FWS-P1A in accordance with N2-OP-3, which placed it in minimum flow. The pump was not immediately loaded due to a precaution in Operating Procedure N2-OP-3, which precludes loading until its lube oil temperature is greater than 130 degrees Fahrenheit. Loading of the Feedwater pump is accomplished by opening its associated flow control valve to the reactor vessel. Even unloaded, there is 8,250 gallons per minute (gpm) pump flow through the minimum flow control valve to the Condenser.

I. DESCRIPTION OF EVENT (cont.)

August 23, 1992

o At 0322 hours, Feedwater Heater Drain pump 2HDL-P1A tripped on low water level in the fourth point heater. This was most probably caused by the Feedwater Heater Drain System controls being less stable at lower power levels.

o At 0325 hours, a second Heater Drain pump, 2HDL-P1C, tripped on low level in the associated fourth point heater. Again, this was most likely caused by instability at lower power levels. At this time the Station Shift Supervisor (SSS) considered loading Feedwater pump 2FWS-P1A, however, the lube oil temperatures had not yet reached 130 degrees Fahrenheit. The SSS then gave the order to unload Feedwater pump 2FWS-P1C by shutting its flow control valve (2FWS-LV10C). This caused 2FWS-P1C minimum flow control valve to open, allowing a nominal 8,250 gpm Feedwater flow to the Condenser. Flow from Feedwater pump 2FWS-P1B then rose to meet the reactor feed demand (it essentially doubled to 14,500 gpm). Operators also began to notice variations in hotwell level indication due to excessive Condensate System flow caused by increased Feedwater demand.

o With 2FWS-LV10C fully shut, Feedwater pumps 2FWS-P1A and 2FWS-P1C were each pumping 8,250 gpm of minimum flow back to the Condenser while Feedwater pump 2FWS-P1B was carrying the system load to the reactor (approximately 14,500 gpm). This represented a load of approximately 30,000 gpm on the Condensate System, which is well beyond the system's capability and therefore overloaded the Condensate System.

o At 0332 hours, Condensate booster pump 2CNM-P2C tripped on low suction pressure.

o At 0333 hours, Feedwater pump 2FWS-P1C was manually tripped and seconds after, Feedwater pump 2FWS-P1B tripped automatically on low suction pressure. With the loss of Feedwater flow from 2FWS-P1B to the reactor, water level quickly dropped to Level 3 (159.3 inches), causing a reactor scram to occur. At Level 3 a Group 4 Primary Containment isolation also occurred.

II. CAUSE OF EVENT

A root cause analysis has been completed in accordance with Nuclear interfacing Procedure NIP-ECA-02, "Root Cause Evaluation."

The root cause for the reactor scram was determined to be personnel error due to an extraneous act. Specifically, the SSS performed an action that was not required by procedure, nor was it in accordance with operator training. At the time the second Feedwater Heater Drain pump unexpectedly tripped, the SSS perceived an urgency to reduce FWS flow.

II. CAUSE OF EVENT (cont.)

The reason for reducing FWS flow was that the SSS felt the loss of Feedwater Heater Drain pumps would cause an increase in the load on the Condensate System. At higher power levels the Feedwater Heater Drain pumps supply approximately one-third of the condensate flow to the FWS pump suctions. However, at 55 percent reactor power the contribution of the Feedwater Heater Drain System is insignificant.

The SSS wanted to continue the FWS pump shift procedure, which required loading 2FWS-P1A while unloading 2FWS-P1C, and then stopping 2FWS-P1C. A precaution in Operating Procedure N2-OP-3 prevented opening the flow control valve for 2FWS-P1A because its lube oil temperature was less than 130 degrees Fahrenheit. Because the SSS was unable to immediately load 2FWS-P1A, he decided to unload 2FWS-P1C. At the time the SSS wrongly assumed that shutting the flow control for 2FWS-P1C would reduce FWS flow demand on the Condensate System. However, the affect of unloading

2FWS-P1C was to increase FWS flow demand by approximately 8,250 gpm. The perceived urgency and inaccurate assessment of the sequence of events that would follow the unloading of a Feedwater pump caused the overload of the Condensate System. The excess Condensate System flow caused a low suction pressure trip of Condensate booster pump 2CNM-P2C, which caused the low suction pressure trip of Feedwater pump 2FWS-P1B.

A contributing factor to this event was the instability of the Feedwater Heater Drain System at lower reactor power. Previously this evolution was performed at 70 percent reactor power, but the power level specified in the Operating Procedure for this evolution was changed earlier this year. At 55 percent reactor power the Heater Drain pumps are more susceptible to tripping than at 70 percent reactor power. A second contributing factor was the precaution in Operating Procedure N2-OP-3, preventing loading a Feedwater pump until its lube oil temperature is greater than 130 degrees Fahrenheit. Subsequent questioning of the vendor determined that loading a Feedwater pump below 130 degrees Fahrenheit does not adversely affect the pump.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10 CFR 50.73 (a)(2)(iv), "any event or condition that resulted in manual or automatic actuation of an Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)." The reactor scram was an automatic ESF (RPS) actuation.

The reactor scram resulted from a Level 3 low reactor water level trip signal that occurred due to a loss of Feedwater System flow. The loss of Feedwater, the reactor scram, the Group 4 Primary Containment isolation, and all events, actions and parameters that followed, were bounded by the analysis for the "Loss of Feedwater Flow" event as discussed in the NMP2 Updated Safety Analysis Report (USAR), Section 15.2.7. Since reactor vessel water level never dropped below 130 inches, no Emergency Core Cooling Systems actuated and level was

III. ANALYSIS OF EVENT (cont.)

restored using Feedwater pump 2FWS-P1A.

The reactor scram on low reactor vessel water level occurred as designed and is required to mitigate the consequences of a Loss of Coolant Accident (LOCA). The Group 4 Primary Containment isolation was a conservative plant response with no operating impact. The reactor scram and the Group 4 isolation in no way impacted the public health and safety nor did they affect the operators' ability to maintain the reactor in a safe condition.

From the time the reactor automatically scrammed until the low reactor vessel water level (Level 3) trip was clear was approximately 20 minutes.

IV. CORRECTIVE ACTIONS

immediate operator actions involved restoring reactor vessel water inventory and commencing a controlled plant shutdown.

Additional corrective actions include:

1. The Operations Training Department will present the details of this event as part of operator training. Also, there will be an initial remedial training using the simulator to demonstrate FWS and CNM response to various transients.

2. The Control Room simulator model will be analyzed and, if necessary, modified to more accurately depict actual FWS and CNM response. 3. Condensate System lesson plans for both initial license and requalification training will be upgraded.

4. The licensed operators involved with this event will write and issue a Lessons Learned Transmittal (LLT) to document the event and its consequences. The LLT will be issued to all site Operations Department personnel by September 30, 1992.

5. Senior Reactor Operators involved with the event have been counseled by the Operations Manager.

6. A Deviation/Event Report (DER #2-92-3313) was issued to prompt an Engineering review of operating with three Feedwater pumps feeding the reactor vessel. This section of N2-OP-3 has been deleted until the disposition of this DER.

IV. CORRECTIVE ACTIONS (cont.)

7. N2-OP-3 has also been revised, removing the 130 degree Fahrenheit lube oil temperature limit for loading Feedwater pumps. In accordance with the vendor's recommendation, the Feedwater pumps are allowed to be loaded whenever lube oil temperature is greater than 50 degrees Fahrenheit.

8. Procedural guidance has been issued to place the Feedwater Heater Drain pumps in a manual minimum flow configuration whenever reactor power is below 65 percent of rated. This will increase the stability of the pumps at lower reactor power levels.

V. ADDITIONAL INFORMATION

A. Failed components: none.

B. Previous similar events:

LER 91-023 describes a similar event where low suction pressure caused a loss of Feedwater pumps and a reactor scram on low water level. LER 91-023 was due to personnel error involving poor communications between Control Room operators. Although the event described in this LER was a personnel error due to an extraneous act, there is a similarity between the two events. in both cases the operators demonstrated a lack of detailed knowledge of the Condensate System. There were no actions to correct the knowledge deficiency in LER 91 -023, because at the time it was determined to be limited to a small sample of licensed operators. Had there been actions to strengthen the operators' Condensate System knowledge, this event may have been prevented.

C. Identification of components referred to in this LER:

Table Omitted.

ATTACHMENT 1 TO 9209300077 PAGE 1 OF 1

NIAGARA
MOHAWK

NINE MILE POINT NUCLEAR STATION/P.O. BOX 32, LYCOMING, N.Y 13093/TELEPHONE (315) 349-2447

Neil S. "Buzz" Carns
Vice President September 21, 1992
Nuclear Generation NMP87286

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 92-17

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

LER 92-17 Is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv), "any event or condition that resulted in manual or automatic actuation of an Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

A 10 CFR 50.72 (b)(2)(ii) report was made at 0652 on August 22, 1992.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,

Mr. N. S. Carns
Vice President - Nuclear Generation

NSC/RLM/lmc
ATTACHMENT

pc: Thomas T. Martin, Regional Administrator Region I
Wayne L. Schmidt, Senior Resident Inspector

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